Approved For Release 2009/08/26 : CIA-RDP88-00904R000100110

Third United Nations
International Conference
on the Peaceful Uses
of Atomic Energy

A/CONF.28/P/585
USSR
May 1964
Original: RUSSIAN

Confidential until official release during Conference

PHYSICAL STUDIES OF NOVO-VORONEZH ATOMIC POWER STATION REACTOR (WWER)

L.V.Komissarov, G.L.Lunin, A.N.Novikov, V.A.Sidorenko, V.D.Sidorenko

I. Core Design

The first unit of Novo-Voronezh Atomic Power Station was described repeatedly in literature [I, 2 , 3]. Thermal power of this water-moderated water-cooled reactor is 760 MW, which corresponds to electric power of 2I0 MW.

The core (diameter ~ 3 m, height ~ 2,5 m) consists of 349 hexahedral assemblies (key dimension - I44 mm, pitch - I47 mm) 3I2 of them are the fuel assemblies, and the rest 37 are movable control assemblies.

The fuel assembly consists of 90 cylindrical fuel elements (external diameter I0.2 mm, hexagonal lattice pitch - I4.3 mm). The fuel elements are filled in by pellets of sintered uranium dioxide. The pellets diameter is 8,7 mm, density - 9.9 - I0.1 gm/cm^3. Zirconium cladding alloyed with niobium [3] has thickness of 0.6 mm. Along an axis of the assembly a zirconium alloy tube was arranged as constructive junction of a spacing system. During experiments these tubes were used as channels for neutron detectors. The external hexahedral jacket of the fuel assembly made of zirconium alloy has a wall thickness of 2.1mm.

The reactor has two kinds of the control assemblies: to compensate slow variations of reactivity (compensating assemblies) and to shut down a chain reaction (safety assemblies). Upper part of the compensating assembly is a hexahedral tube absorber with the "black" walls for thermal neutrons. At operating conditions this tube is filled in with water what made it efficient not only to thermal, but to fast neutrons as well. The lower parts of the compensating assemblies are almost completely similar to the fuel ones. Therefore moving of the compensating assemblies leads to an absorber-fuel replacement in the core.

The safety assembly consists of a guide tube, an absorber and its follower. The zirconium alloy guide tube hexahedral outside and cylindrical inside is mounted steadily in the core. Inside of the tube a mobile cylindrical part of the safety assembly (absorber and its follower) is arranged. The absorber is made as filled with water cylindrical tube

of external diameter I25 mm with "black" walls for thermal neutrons. The follower of the safety assembly is a cylindrical zirconium alloy scatterer without fuel with longitudinal water channels.

A full number of 37 control assemblies comprises 6 safety and 3I compensating assemblies. They are uniformly arranged amidst the fuel assemblies of the central part of the core with mean ratio one to six (389 mm pitch).

The core design put a series of neutron and engineering problems therefore considerable amount of critical experiments and also methodical and calculating investigations have been done. Many of them are usual for any large power water-moderated water-cooled reactor with low enrichment oxide fuel. Some problems, however, were found specific ones because of peculiarities of the designed core. In particular, essentially high efficiency is characteristic for the control rods described. Therefore comparatively few number of the control rods is sufficient on principle for governing the reactor without any additional methods of reactivity compensation. Such controls, however, lead to considerable disturbances of the neutron and heat distribution in the core. Besides, the efficiency of any group of control assemblies appears to be strongly dependent on the amount and location of the rest ones. Therefore especially careful study of reactor operational features was necessary for rational choice of amount and location of the safety assemblies, division of the compensating assemblies into operating groups, their lifting succession etc.

It is intended the reactor to be exploited at stationary regime with periodical partial reloadings [4] the core always consisting of assemblies with different fuel burn-up. To simplify achievement of such a regime the first fuel charge of the reactor was adopted non-uniform. The first core consists of assemblies with uranium of 2,0% and I,5% enrichment and with uranium of natural composition.

2. Cold Lattice Characteristics

A practice of reactor core design has a deal with a number of different and ruther cumbersome computing tasks. It is very important to have a simple and effective calculating method, which taking into account the main physical peculiarities of the core secure practically sufficient accuracy. As for large uranium-water reactors, where neutron spectrum is not very hard, reflector action is not essential and neutron leakage occurs at first flight and during slowing down mainly it is

585

possible in our opinion, to use of one group diffusion form of criticality condition, epithermal reactions having been taken into account effectively [4 , 5]. This conclusion follows from an analysis of uranium-water lattices experimental data [4 , 6] and results of experiments with WWER fuel assemblies particularly [7].

In experiments [7] the multiplicative parameters of uniform (I,5% and 2,0% uranium enrichment and natural uranium) and of "mixed" fuel lattices were studied. The uniform and "mixed" lattices were composed of WWER fuel assemblies in different combinations.

Investigations with 1.5% and 2.0% enrichment have been performed by "supercritical method" to measure the multiplicative parameters and verify the theoretical assumptions.

In one-group approximation, which is justified in respect to the uranium-water lattices with not very hard neutron spectrum, and $L^2 << \tau$ the criticality condition is to be

$$K_{ef} = K_{\infty} \tilde{F}(B^2M^2) = I$$

$\tilde{F}(B^2M^2)$ - Fourier-image for the point fission kernel,
B^2 - geometrical buckling
M^2 - migration area

Let the critical state of the assembly correspond to the geometrical buckling $B^2 = B_0^2$. Then a derivative function of reactivity near criticality is

$$\left(\frac{d\rho}{dB^2}\right)_{B_0^2} = \left(\frac{d \ln \tilde{F}(B^2M^2)}{dB^2}\right)_{B_0^2}$$

For $\tilde{F}(B^2M^2)$ we shall use Komissarovis empiric ecpression

$$\tilde{F}(B^2M^2) = \left(I + \frac{B^2M^2}{n}\right)^{-n}$$

When n=l and $n \to \infty$ it gives leakage escape probability in diffusion and age approximation respectively, and when $n \simeq 0.38$ - approximatly in transport representation. Then

$$-\left(\frac{d\rho}{dB^2}\right)^{-1}_{B_0^2} = \frac{1}{M^2} + \frac{B_0^2}{n}$$

Thus, dependence of experimentally obtained values $(d\rho/dB^2)_{B_0^2}$ on B_0^2 contains the information not only of the fuel lattice migration area

$$\frac{1}{M^2} = -\lim_{B_0^2 \to 0}\left(\frac{d\rho}{dB^2}\right)^{-}_{B_0^2}$$

but also of neutron leakage caracter "n"

$$\frac{1}{n} = -\frac{d}{dB_0^2}\left(\frac{d\rho}{dB^2}\right)^{-1}_{B_0^2}$$

585

The values $(d\rho/dB^2)_{B_0^2}$ precisely measured in experiments described show the linear dependence up to $B_0^2 = 60\ m^{-2}$. The least squares treatment of data gives $M^2 = 49.7 \pm 0.9\ cm^2$ $n = 1.04 \pm 0.08$.

The equality of n to unit within the experimental errors justified the use of diffusion approximation and the notions of "multiplicative coefficient" K_∞ and "material parameter" $\varkappa^2 = \frac{K_\infty - 1}{M^2}$ provided the equation of criticality to be $\varkappa^2 = B_0^2$.

An objective of "mixed" lattice experiments was to verify the average procedure the multiplicative parameters concerned and to evaluate the irregularities of heat gene... ng in the reactor with periodically non-uniform structure. Critical... ty of the core in the experiments was adjusted by change of the moderator level. The measured multiplicative parameters of the lattices are compared with calculated ones in Table I.

The effect of epithermal reactions in the calculation of the multiplicative parameters was account by method, described in [4]. The transfer of thermal neutrons between the assemblies of different kind was taken into account in the "mixed" lattices calculation. The calculated data of Table I was obtained before performance of the experiments.

Fig. 1 presents the results of the measurements for "mixed" lattice, composed with assemblies of natural uranium and of 1.5% enrichment in 2:1 ratio . The neutron density ratio $\frac{\phi\ 0.714\%}{\phi\ 1.5\%}$ was measured by small fission chamber (U-235). As one could see this ratio keeps constant meaning along the radius of the core and equals in average 1.31 ± 0.06.

Fig. 2 shows the results of measurements of neutron densities along the assembly radius. Surrounding as... mblies in this experiment were of the same kind (1.5% uranium enrichment). The measurements were accomplished by method of fuel element activation and controlled by registration of high energy gamma-guants to except the activation of zirconium-alloy coating.

3. Physics of WWER Control Rods

Efficiency of the absorbers in WWER is about 70% comparably with "black" absorber because of the large dimensions and the presence of the moderator inside. It depends rather slightly on L^2 of the lattice and on "blackness" of absorber walls for the thermal neutrons because of $L^2 \ll \tau$ Particularly, the efficiency of water cavity having the dimensions of WWER absorber is more than 60% of its efficiency according to [8].

As one could expect, the diffusion approximation is reasonable one for calculating the absorber efficiency, provided the mutual distances between the absorbers and to the external boundary exceed some migration lengths. Table II gives the comparison of ecperimental and calculated data the absorber efficiency concerned. The experiment was fulfilled with the critical core, composed of 85 1.5% - enrichment assemblies [8] . Absorbers were placed instead of the fuel assemblies simmetrically around the core axis. Criticality of the systems was adjusted by the level change of the moderator. The calculation was made by method [9] in two-group diffusion approximation, the code constituted by V.I.Lebedev.

Fig.3 shows the distribution of thermal neutron flux for the composition of 8I assemblies (1.5% enrichment) and 4 absorbers. The measurement was made by small fission chamber U-235. Results with 5% uncertainties are given in comparison with calculated data (solid curve) Surface of the core and absorber boundaries were supposed cylindrical in calculation.

As follows from the comparison of the experimental and calculated data concerned the efficiency of the water cavity [8] , the one-group calculation may be also quite satisfactory, provided the reasonable choice of the effective boundary conditions for the absorbers. The reason is that the noticeable disturbances, caused by the absorbers, reveal itself in the region of the small statistics weights (within the distance of order ~L from the surface of absorbers).

In one-group theory the insertion of the "black" absorbers into the core may be treated as the change of its geometrical buckling. This conclusion was confirmed experimentally by results that derivative of the assembly reactivity as function of moderator level $d\varrho/dh$ within the experimental errors is independent on the presence of absorbers in the core. In common case the one-group theory gives:

$$\frac{d\varrho}{dh} = \frac{2\pi^2}{h^3}\,\frac{\int \mathcal{D} j^2 dv}{\int K_\infty \xi\Sigma_s j^2 dv} = \frac{2\pi^2}{h^3}\,\frac{\overline{\mathcal{D}}}{(\overline{K_\infty \xi\Sigma_s})}$$

where D is diffusion coefficient, $\xi\Sigma_s$ is slowing-down cross-section, j is slowing-down density. This means that the mean characteristics of the core are defined by fuel lattice characteristics:

$$\frac{\overline{\mathcal{D}}}{(\overline{K_\infty \xi\Sigma_s})} \simeq \frac{M^2}{K_\infty}$$

or (as K_∞ for absorbers equals to 0 wittingly) $\overline{K_\infty} = K_\infty$ and $\overline{\mathcal{D}}/\overline{\xi\Sigma_s} = M^2$.

585

The calculation shows, that logarithmic derivative of the neutron density at the absorber surface (and therefore effective geometrical buckling of the core with the absorbers) rather slightly depends upon the reactor temperature. For this reason one could assume the efficiency of the absorber to be risen as $\frac{M^2}{K_\infty}$ of fuel lattice, while the reactor heating.

The model for calculation, treating the absorber insertion into the core as a change of its geometrical parameter is available to analyse efficiency of partly removed absorvers, effects of xenon poisoning etc. In this model the calculation of efficiency for partly removed absorbers is reduced to finding the effective geometrical buckling B^2 of the core as function of absorber position $h_\ell \ (0 \le h_\ell \le H)$

$$\Delta\phi + B^2\phi = 0 \qquad \Delta\rho(h_\ell) = \frac{\Delta B^2 M^2}{K_\infty} \tag{1}$$

Using the approximate separation of variables

$$\phi = \begin{cases} Z_1(h)\cdot\varphi_1(r,\varphi) & 0 \le h \le h_\ell \\ Z_2(h)\cdot\varphi_2(r,\varphi) & h_\ell \le h \le H \end{cases} \tag{2}$$

where φ_1, and φ_2 represent the asymptotic radial-asimutal distributions of the neutron flux for lower and upper sections of the core respectively, $\Delta\varphi_i + B^2_{r\varphi i}\varphi_i = 0$, the function $B^2(h)$ can be found by the solution of the corresponding equations for the slab:

$$\Delta Z_1 + (B^2 - B^2_{r\varphi_1})Z_1 = 0 \qquad Z_1(h_\ell) = Z_2(h_\ell)$$

$$\Delta Z_2 + (B^2 - B^2_{r\varphi_2})Z_2 = 0 \qquad \dot{Z}_1(h_\ell) = \dot{Z}_2(h_\ell) \quad Z_1(0) = Z_2(H) = 0$$

The results of calculation are represented in dimensionless form in Fig.5 and Fig. 6. Corresponding change of nonuniformity factor for the axial neutron flux is shown in Fig.7.

Fig.8 shows the example of calculation for the case of one central absorber removed. The differential absorber efficiency $\Delta\rho(h_\ell)$ and volume nonuniformity factor $K_V = \phi_{max}/\bar{\phi}$ were obtained by solving the equation

$$\Delta\phi + \frac{K_\infty/K_{ef} - 1}{M^2}\phi = 0$$

by method of approximate separation of variables and by method of synthesis [IO]

$$\phi = \begin{cases} Z_{11}\varphi_1 + Z_{12}\varphi_2 & 0 \le h \le h_\ell \\ Z_{21}\varphi_1 + Z_{22}\varphi_2 & h_\ell \le h \le H \end{cases} \tag{3}$$

585

When the diffusion coefficients of fuel lattice and absorber are equal, the synthetic method gives accurate solution of equation (1), provided φ_1 , and φ_2 to be obtained from equations:

$$\Delta \varphi_i + \left[\frac{K_\infty / K_{ef\,i} - 1}{M^2} - \left(\frac{\pi}{H}\right)^2\right] \varphi_i = 0 \qquad i = 1,2$$

where K_{ef1} and K_{ef2} are effective multiplicative factors for extreme upper and lower position of removable absorber group. In Fig.8 the diffusion coefficient of absorber equals to that of fuel lattice. Comparing the results one may conclude that the method of approcimate separation of variables gives rather satisfactory description of $\Delta\rho(h_6)$ but can give a noticeable overestimation of volume nonuniformity factor $K_v = \Phi_{max}/\bar{\Phi}$

It may be pointed out that functions $\Delta\rho$ and $\frac{d\rho}{dh}$ (Fig.5 and Fig.6) can be used to evaluate subcriticality of the core by lifting of absorbers, for this functions depend on the whole efficiency of the absorber group only. Indeed, if criticality of the core may be reached by the lifting of an absorber group, the measurements of "h_6" and $\left(\frac{d\rho}{dh}\right)_{h_6}$ for this group are sufficient to have the evaluation.

Justification of the approcimation used was indirectly confirmed by the measurement of temperature reactivity coefficient ("black absorber shifting" method). The experiments were carred out on two critical cores filled in fully with water. The first core consisted of 78 fuel assemblies (1.5% enrichment) and 7 absorbers of compensating assemblies. The second core comprised I32 assemblies and I9 absorbers. The absorbers were placed uniformly amidst the fuel assemblies in both cores. The central absorber was removable and supplied by fuel follower with uranium of 1.5% enrichment. The temperature dependence of the critical position for the central absorber and of its $\partial\rho/\partial h$ value were measured.

According to abovementioned assumptions the criticality condition $\rho(h_6)=0$ may be expressed as $B^2(h_6(t),t) = \varkappa^2(t)$ where $\varkappa^2$ is material buckling of the fuel lattice.

Derivation of this equalities, taking into account $\partial\rho/\partial h = -\frac{M^2}{K_\infty}\frac{\partial B^2}{\partial h}$ gives:

$$\frac{\partial\rho}{\partial t} = -\frac{\partial\rho}{\partial h}\cdot\frac{dh_6}{dt} = \frac{M^2}{K_\infty}\left(\frac{d\varkappa^2}{dt} - \frac{\partial B^2}{\partial t}\right)$$

Inspite of essential differences of the core dimensions and compositions the temperature dependence for both series of measurements was found to be the same within the experimental uncertainties. According to calculation for both experimental cores term $\frac{M^2}{K_\infty}\frac{\partial B^2}{\partial t}$ weakly depends on temperature and contributes to $\partial\rho/\partial t$ less than I0%.

585

Therefore, both series of measurements give a similar temperature dependence for temperature coefficient of fuel lattice $\frac{d\rho}{dt} = \frac{M^2}{K_\infty}\frac{d\varkappa^2}{dt}$ Results of measurements for the first core are compared with the calculated data in Table III.

The calculation of $d\rho/dt$ was made by method described in section 5, for calculation of $\partial\rho/\partial h$ the method of approximate separation of variables was used.

4. Exploitation Characteristics of Control Rods Syst[illegible]

Rational choice of amount and location of the safety assemblies, division of the compensating assemblies into operative groups, determination of their permissible speeds and shifting seccession were rather difficult tasks for the designed WWER control rods system. It was necessary the following requirements to be satisfied simultaneously at any state of the core:

1) Compensating and safety assembly systems have to conform to "Nuclear Safety Standards for the Power Reactors in USSR" [II];

2) The core heat distribution has to permit reactor running at fuel power.

3) Disturbances of the neutron flux while a sharp moving of an operative absorber group have to be the smallest (to avoid the consequent swinging due to Xe-I35 redistribution).

When choicing of the safety assembly locations and of compensating groups lifting succession the most unif[illegible] fuel burnup (to have long core life) was keeped in mind too.

After computing and analysing more than 300 critical situations a control rods system with six safety assemblies was adopted. The arrangement of the system in the core and division of control rods into operative groups are shown in Fig.9 (compensating groups marked according to their lifting succession). There is an interlock preventing lift of any compensating assembly before at least three safety assemblies to be lifted. The speeds of the compensating assemblies at manual control are ~ 1 cm/sec. The speed interval at automatic control (K_1 and K_2 groups) is within 0 to 0.9 cm/sec.

The typical example of the thermal neutron flux distribution in the core in pesence of absorbers and of safety scatterers is shown in Fig.4.

Table IV represents the calculated results of effective geometrical buckling B^2, the efficiency of safety assemblies $\Delta\rho_A$ and of operative compensating groups $\Delta\rho_K$ and of thermal neutron flux nonuniformity factor in plane K_r. The calculation is made in assumption of uniform fuel enrichment and of cylindrical boundary of the core. Efficiencies $\Delta\rho_A$ and $\Delta\rho_K$ are not correlated with the criticality of the reactor (the values M^2/K_∞ were taken constant and equal to 43.5 and 60.0 cm^2 for cold and operating conditions respectively, independently of amount of absorbers in the core). The comparison of B^2 for cold and operating conditions gives an information on temperature influence of boundary effects.

In accordance with "Safety Standards" [II] a special interlock system is foreseen to prevent the possibility of building up supercriticality $\Delta\rho \geq \beta$ during manipulation with the absorbers. Injection of liquid absorber into the reactor coolant is provided for a case of the accidental sticking of the control assemblies.

5. Choice and Characteristics of the First Fuel Load

After preliminary analysis it was decided to exploit the reactor at stationary regime of 2% enrichment uranium feeding. At three reloadings per fuel lifetime it is possible in this case to have fuel burnup of I2000 $\frac{MW \cdot d}{t}$ (in presence of burnable poison). The first fuel load (see Fig.IO) imitates the core after one of the reloading just now finished.

The first core comprises I27 assemblies with uranium of 2% enrichment, I86 assemblies with uranium of 1.5% enrichment and 30 assemblies with natural uranium. All of 2% enrichment assemblies have burnable poison – external hexahedral jacket of the assembly contains about 0.07 wt% of natural boron. Initial multiplicative properties of such an assembly are similar to those of 1.5% enrichment one . There is no burnable poison in 2% enrichment follower of the compensating assembly.

At lower position of the control assemblies the dioxide uranium contents in the core is about 39.7 tons. After lifting the compensating assemblies it increases to 43.6 tons. The average initial uranium enrichment of the first fuel load is 1.58% and 1.62% at lower and upper position of the compensating assemblies respectively.

The expected effects of reactivity for the first core are presented in Table V.

The reactivity effects and fuel burnup calculations have been done on electronic computer M-20 of Kurtchatov Atomic Energy Institute. For the neutron spectrum calculation in the range of 0÷0.6 eV the integral transport method was used (heavy gas approximation without taking into account the neutron slowing-down in fuel). The deformation of thermal neutron spectrum caused by fuel burnup is shown in Fig.II. In the region of 0.3 eV the deformation is due to Pu-239 and Pu-24I resonances.

For epithermal spectrum calculation the multigroup diffusion code was used [I2] . The leakage escape probability during neutron slowing down was defined by averaging of corresponding cross sections over this spectrum and taking into account the first flight correction. At multiplicative coefficient calculation the spectrum above 0.6 eV was supposed to be proportionate to $\frac{1}{E}$. Selfprotection of the fuel in epithermal region was considered for isotopes U-238, Pu-240 and Pu-242 only.

Effective resonance integral of U-238 for a single fuel rod was calculated in the narrow-resonance approximation. Temperature dependence of resonance integral was admitted in accordance with [I3] .

Fig. I2 represents the excess reactivity of the core in operative condition and subcriticality of the core in cold condition as function of fuel burn-up. The fuel burning nonuniformity adopted in calculations corresponds to the chosen succession of control assemblies shifting.

When this report was being written, the cold core experiments had been completed. The initial stage of experiments was carred out on the core mockups. The cold critical tests of the real core were accomplished in December I963 directly in the reactor vessel of Novo-Voronezh Atomic Power Station.

Apart of a chosen core study the experimental program comprised the investigation of core characteristics dependence upon fuel assembly composition and location. Nine different core mockup compositions were assembled in all. The influence of the core periphery composition on the efficiency of compensating system was carefully investigated particularly. For the adopted core version series of measurments have been done - of excess reactivity, subcriticality at the lower control rods position,efficiencies of control rod groups and of neutron flux distribution in the core as well.

Results of investigation of the subcriticality, the reactivity excess and the control rod efficiencies at 20°C for the first core are presented in Table VI. The core criticality in the experiments was obtained by change of the moderator level. The excess reactivity and control rod efficiencies were accounted by integration of the measured values $d\rho/dh$. Subcriticality of the core was estimated by suitable treatment of inverse counting rate $\frac{1}{N}$, depending on the moderator level change. The experimental uncertainties are ± 0.005 for full excess reactivity and ± 0.0025 for subcriticality.

The full excess reactivity was measured also by poisoning with boron acid solution. The result 0.114±0.005 is quite compatible with the given in Table VI.

Experimental data (Table VI) are in a good agreement with the calculated results (see Tables IV, V and Fig.I2). Discrepancies for the efficiencies of compensating system (0.141 ± 0.007 measured and 0.127 calculated) and of separate operative groups are caused by core nonuniformity, neglecting in calculation. Results of neutron flux measurments are being now in stage of treatment.

The big collective of collaborators took direct participation in this work: G.Y. Andrianov, V.A. Voznesenski, V.S. Ionov, A.N.Kamishan, V.A. Kuzmichova, D.M. Petrunin, V.N. Semionov, V.I.Halizev and mathematician group: V.I.Lebedev, A.I.Kuleshov, E.D. Beljaeva. Authors express special acknowledgements to S.A. Skvortsov and N.A. Lazukov for guidance participation.

585

References

I. Project of Atomic Power Station on 420 MW(e). Report No.8, International Power Conference, Belgrad (I957)

2. S.A. Skvortsov - Water-Moderated Water-Cooled Reactors in USSR, Report No.2184, The Second International Conference on the Peaceful Uses of Atomic Energy, Geneva (I958)

3. R.S. Ambartsumjan et al. Fuel Elements for Water-Moderated Water-Cooled Power Reactors. Paper No2196.The Second International Conference on the Peaceful Uses of Atomic Energy, Geneva (1958)

4. S.M.Feinberg et al. - Fuel Burn-up in Water-Moderated Water-Cooled Power Reactors and Uranium-Water Lattice Experiments.Paper No2145, The Second International Conference on the Peaceful Use of Atomic Energy, Geneva (I958)

5. S.M.Feinberg On the Uranium-Water Lattice Theory, USSR Science Academy Conference on the Peaceful Use of Atomic Energy.(Section of Physics and Mathematics) Moscow (I955)

6. L.G.Andreev et al. - Physical Parameters of Uranium-Water Lattices. The Third International Conference on the Peaceful Use of Atomic Energy, Geneva (I964), P/358.

7. G.Y.Andrianov et al. WWER Core Investigation. Symposium on Exponential and Critical Assemblies,Amsterdam (I963)

8. V.N. Semionov et al. Efficiency of WWER Control Rods. Symposium on Physics and Materials of Absorbers. Vienna (I963)

9. G.V.Sinjutin, V.N. Semionov - "Efficiency of Control Rods in Cylindrical Reactor". Report No2469. The Second International Conference on the Peaceful Uses of Atomic Energy, Geneva (I958)

IO. S.Kaplan - Some New Methods of Flux Synthesis, Nucl.Sci and Eng. I3, 22 (I962)

II. G.A. Gladkov et al. Nuclear Safety Standards for Power Reactors in the USSR. The Third International Conference on the Peaceful Uses of Atomic Energy, Geneva (I964) P/380.

I2. G.I. Martchuk - Reactor Calculation Methods, Gosatomizdat (I96I)

I3. L.W. Nordheim - A New Calculation of Resonance Integrals, Nucl. Sci. and Eng. I2, 467 (I962)

585

Table I.

Multiplicative Prameters of WWER Cold Fuel Lattices

Enrichment	M^2 cm^2 experiment	calculation	$\varkappa^2$ m^{-2} experiment	calculation	K_∞ experiment	calculation
2.0		47.4	57±0.7	58	1.283±0.007	1.281
1.5		47.9	36.1±0.5	34.5	1.179+0.004	1.172
0.714		49.3	-25.9±0.3	-33	0.867	0.847
2+1.5(1:1)	49.7±0.9	47.6	47.1±0.5	-	1.234±0.005	1.230
2+1.5(1:2)		47.7	43.4±0.5	-	1.216±0.005	1.215
2+0.7(1:1)		-	18.2±0.1	-	1.090±0.007	1.100
1.5+0.7(2:1)		-	16.7±0.1	-	1.083±0.005	1.088

Table III

Results of Temperature Measurements with Fuel Lattice of 1.5% Enrichment

t	Experiment				Calculation	
	h_e	$\frac{\partial\rho}{\partial h}\cdot 10^5$	$\frac{\partial\rho}{\partial t}\cdot 10^4$	$\frac{d\rho}{dt}\cdot 10^4$	$\frac{d\rho}{dt}\cdot 10^4$	$\frac{\partial\rho}{\partial h}\cdot 10^5$
°C	cm	cm^{-1}	$°C^{-1}$	$°C^{-1}$	$°C^{-1}$	cm^{-1}
20	97.31	36.2±0.5	1.27±0.08	1.37±0.10	1.32	38.0
40	103.98	30.7±0.5	1.35±0.08	1.44±0.10	1.42	33.5
60	113.92	25.1±0.5	1.44±0.08	1.53±0.10	1.56	27.7
80	126.22	19.5±0.4	1.54±0.08	1.65±0.10	1.74	22.4

585

Table II

Absorber Efficiencies

Number of absorbers in experiment	Raius of absorber locations	Efficiency $\Delta\rho$ 10^2 Experiment	Calculation
1	0	2.47±0.13	2.37
1	14.7	2.05±0.11	1.95
1	25.5	1.43±0.07	1.42
1	29.4	1.31±0.05	1.26
1	38.9	0.85±0.04	0.88
1	44.1	0.71±0.04	0.70
3	14.7	5.29±0.26	5.10
3	25.5	6.30±0.30	6.10
3	29.4	5.90±0.28	5.72
3	38.9	3.84±0.19	3.65
3	44.1	2.76±0.14	2.66
6	25.5	10.20±0.43	9.90
6	38.9	8.90±0.39	8.60
6	44.1	6.10±0.29	5.85

585

585

Table IV

Characteristics of Control Rod System (calculated)

The core comprises absorvers of operative groups	B^2 m^{-2}			$\Delta \varrho_K$		$\Delta \varrho_A$		K_z
	20°C	263°C	Group	20°C	263°C	20°C	263°C	
$A_1A_2K_1-K_8$	35.4	34.6	-	-	-	-	-	1.67
$A_2K_1-K_8$	31.6	30.2	-	-	-	0.0163	0.0267	-
K_1-K_8	30.0	28.4	K_1	0.0316	0.0379	0.0236	0.0371	2.79
K_2-K_8	22.7	22.1	K_2	0.0124	0.0193	0.0265	0.0300	3.52
K_3-K_8	19.8	18.9	K_3	0.0045	0.0051	0.0211	0.0314	2.64
K_4-K_8	18.8	18.0	K_4	0.0136	0.0174	0.0186	0.0291	2.21
K_5-K_8	15.7	15.1	K_5	0.0067	0.0095	0.0127	0.0196	2.27
K_6-K_8	14.2	13.5	K_6	0.0126	0.0161	0.0121	0.0188	1.75
K_7-K_8	11.3	10.9	K_7	0.0068	0.0091	0.0093	0.0142	1.96
K_8	9.7	9.4	K_8	0.0151	0.0209	0.0085	0.0134	1.89
-	6.2	5.9	-	-	-	0.0110	0.0169	1.88

$\Sigma \Delta \varrho (20^\circ C) = 0.1269$ $\Sigma \Delta \varrho (263^\circ C) = 0.1723$

Table V

Effects of Reactivity for the First Core

Reactivity changes	
Temperature effect	0.024
Dppler effect	0.016
Poisoning effect Xe,Sm	0.039
Fuel burn-up	0.035
Full excess reactivity	0.114

Table VI

Core Characteristics at 20°C (measured)

The core comprises absorbers of operative group	The Core Reactivity	Efficiency of operative group: group	$\Delta\rho_K$	Efficiency of the safety assemblies $\Delta\rho_A$
$A_1A_2K_1$-K_8	-0.025	-	-	-
A_2K_1- K_8	-0.005	-	-	0.020
K_1 - K_8	0.0012	K_1	0.0314	0.026
K_2-K_8	0.0326	K_2	0.0154	0.020
K_3- K_8	0.0480	K_3	0.0070	0.020
K_4- K_8	0.0550	K_4	0.158	0.016
K_5 - K_8	0.0708	K_5	0.0080	0.010
K_6 - K_8	0.0788	K_6	0.0124	0.011
K_7 - K_8	0.0912	K_7	0.0066	0.009
K_8	0.0978	K_8	0.0183	0.009
-	0.1161	-	-	0.011

585

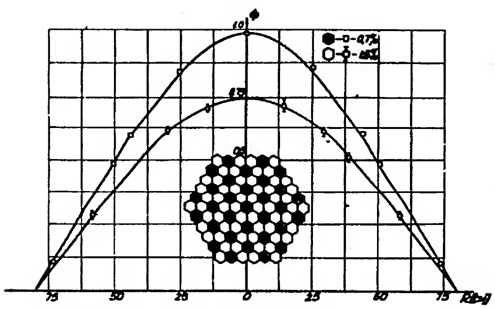
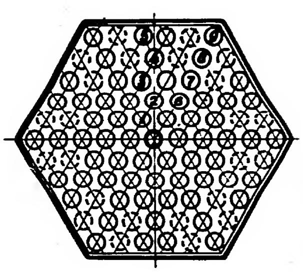


Fig.1. Neutron flux measurement in the "mixed" fuel lattice



№	A
1	1.000
2	0.930
3	0.978
4	0.954
5	1.061
6	0.930
7	0.954
8	0.966
9	1.100
$\frac{\Delta A}{A} = \pm 0.030$	

Fig.2. Neutron flux distribution in the fuel assembly with uranium of 1.5 per cent enrichment

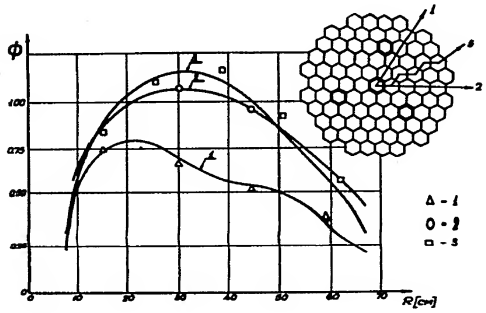


Fig.3. Thermal neutron flux distribution in an experimental care with absorbers

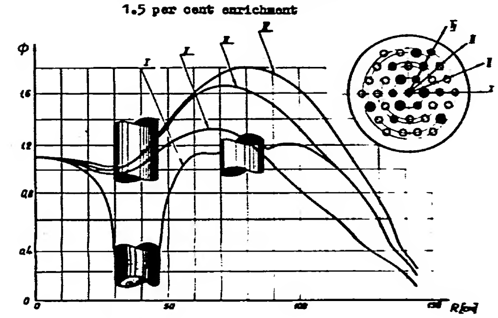


Fig.4. Neutron flux distribution in the uniform core with absorbers of operative group No 8.

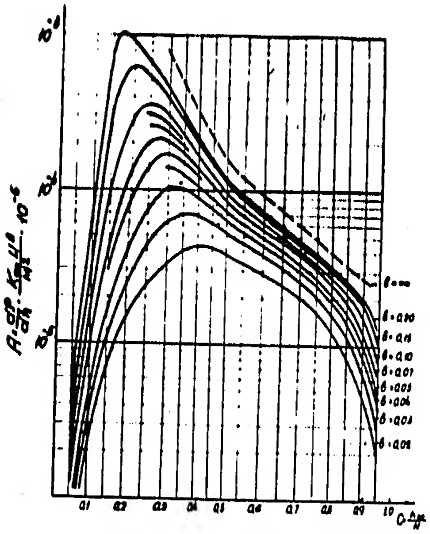


Fig.5. Calculated data on $d\rho/dh$

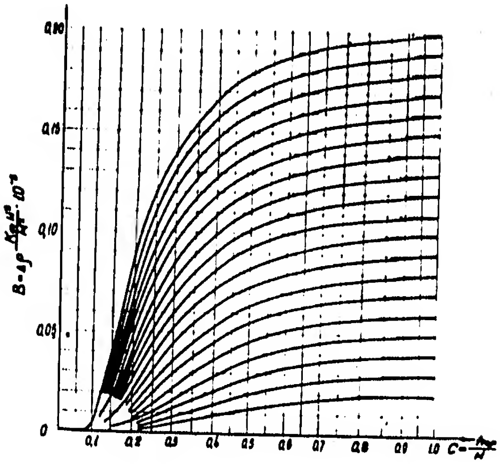


Fig.6. Calculated data on $\Delta\rho$

$$B = (\Delta\rho)_{c=1} \cdot \frac{k_\infty H^2}{M^2} \cdot 10^{-3}$$

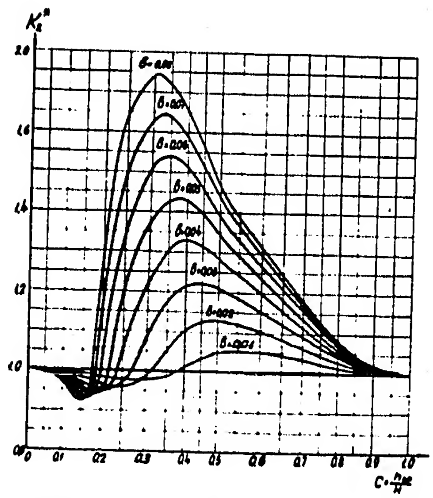


Fig.7. Change of neutron flux nonuniformity factor due to control rod lifting

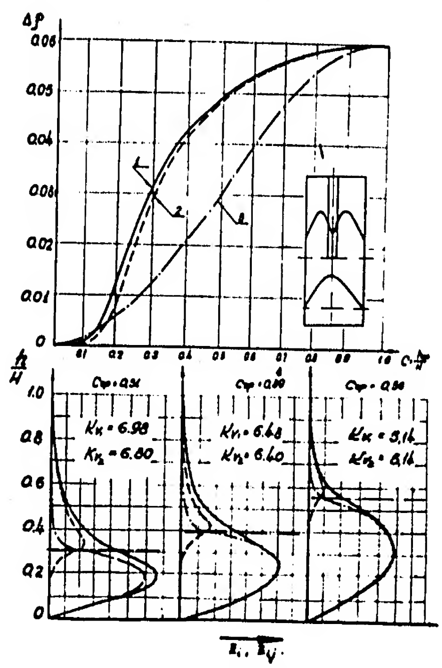


Fig.8. Reactivity ρ and neutron flux nonuniformity factor K_v.
1 – approximate separation of variables, 2 – synthesis method
3 – perturbation theory.

$$K_z^* = \frac{z_{max}}{z} \frac{2}{\pi} \qquad B = (\Delta\rho)_{c=1} \frac{k_\infty H^2}{M^2} \cdot 10^{-3}$$

585

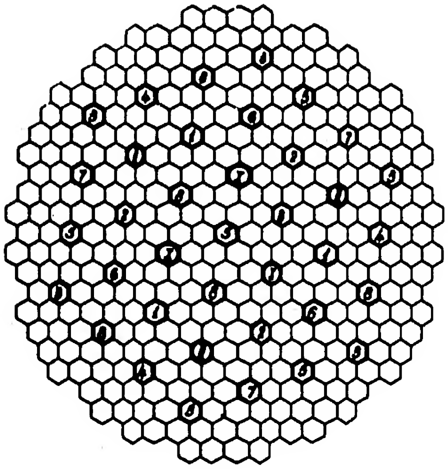


Fig.9. Location of the control rods in the core and their distribution through operative groups.
I - II - safety assemblies,
1 - 8 - compensating assemblies

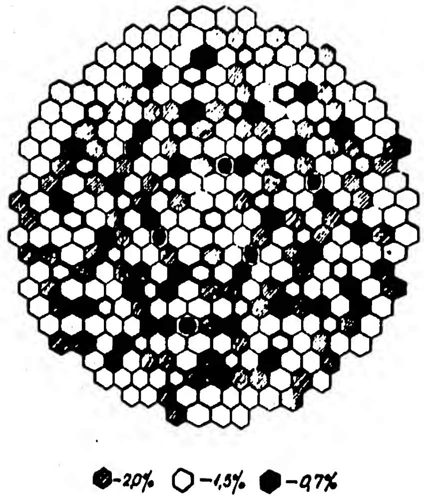


Fig.10. Chart of the first fuel loading

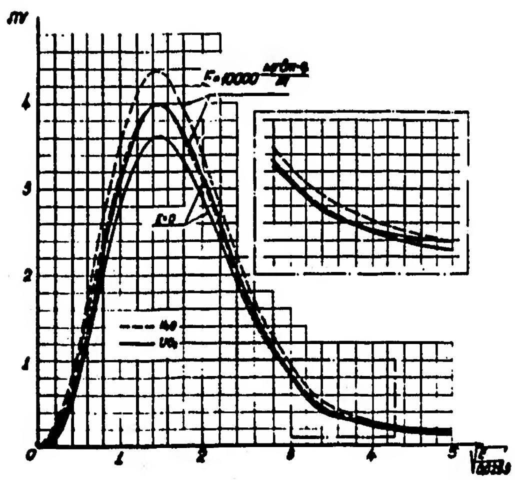


Fig.11. Disturbance of thermal neutron spectrum due to fuel burnup (assembly of 2 per cent enrichment).The spectra are normalized assuming specific power constancy.

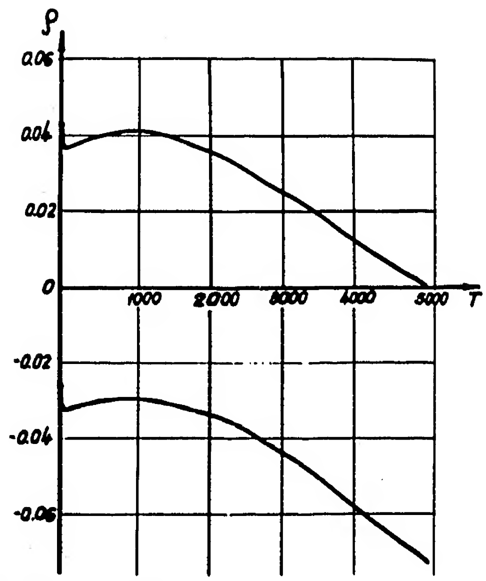


Fig.12. The core excess reactivity and subcriticality vs full power hour operation (operating and cold unpoisoned conditions respectively)

Approved For Release 2009/08/26 : CIA-RDP88-00904R000100110037-9